Fusion Engineering and Design 96–97 (2015) 34–41

Contents lists available at ScienceDirect

Fusion Engineering and Design

journal homepage: www.elsevier.com/locate/fusengdes

Engineering challenges and development of the ITER Blanket System and Divertor

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ARTICLE INFO

Article history:
Received 17 September 2014
Received in revised form 29 May 2015
Accepted 10 June 2015
Available online 23 June 2015

Keywords:
Blanket
Divertor

ABSTRACT

The ITER Blanket System and the Divertor are the main components which directly face the plasma. Being the first physical barrier to the plasma, they have very demanding design requirements, which include accommodating: (1) surface heat flux and neutronic volumetric heating, (2) electromagnetic loads, (3) nuclear shielding function, (4) capability of being assembled and remote-handled, (5) interfaces with other in-vessel components, and (6) high heat flux technologies and complex welded structures in the design. The main functions of the Blanket System have been substantially expanded and it has now also to provide limiting surfaces that define the plasma boundary during startup and shutdown. As regards the Divertor, the ITER Council decided in November 2013 to start the ITER operation with a full-tungsten armour in order to minimize costs and already gain operational experience with tungsten during the non-active phase of the machine. This paper gives an overview of the design and technology qualification of the Blanket System and the Divertor.

1. Introduction

The ITER Blanket System and the Divertor are the main components which directly face the plasma (Fig. 1). Being the first physical barrier to the plasma, they have very demanding design requirements, which include accommodating: (1) surface heat flux and neutronic volumetric heating, (2) electromagnetic loads, (3) nuclear shielding function, (4) capability of being assembled and remote-handled, (5) interfaces with other in-vessel components, e.g. diagnostics, in-vessel coils, cabling, and (6) high heat flux technologies and complex welded structures in the design.

This paper summarizes the main requirements, the design and the related R&D and technology qualification of the Blanket System and the Divertor. An overview of the integration requirements and the remote handling design and R&D can be found elsewhere [1–3].

2. The Blanket System

2.1. Overview

The main functions of the Blanket System are to:

- Constitute the primary interface to the plasma in the main chamber providing a plasma-facing surface compatible with the plasma performance requirements (heat loads, impurity influx) and a limiting surface defining the plasma boundary during limiter operation and plasma start-up/ramp-down.
- Contribute in providing neutronic shielding to the Vacuum Vessel (VV) and external vessel components.
- Contribute in absorbing radiation and particle heat fluxes from the plasma and from the plasma heating systems.
- Provide heat and neutronic shielding to in-vessel diagnostics, such as waveguides, bolometers and in-vessel coils.

Provide passage for the plasma diagnostics, viewing systems, microwave antennas or launchers, neutral beam injectors, the gas and pellet fuelling systems and other minor ancillaries.

The Blanket System consists of 440 Blanket Modules (BM) covering ~600 m^2 as illustrated in Fig. 2. A BM comprises two major components: a plasma facing First Wall (FW) panel and a Shield Block (SB). Each BM is attached to the VV through a mechanical attachment system of flexible supports and a system of key pads. Each BM has electrical straps providing electrical connection to the VV to route halo currents appropriately. Cooling water is supplied to the BM by manifolds supported on the VV behind or to the side of the SB and is designed to remove up to 736 MW of thermal power

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http://dx.doi.org/10.1016/j.fusengdes.2015.06.045
0920-3796/

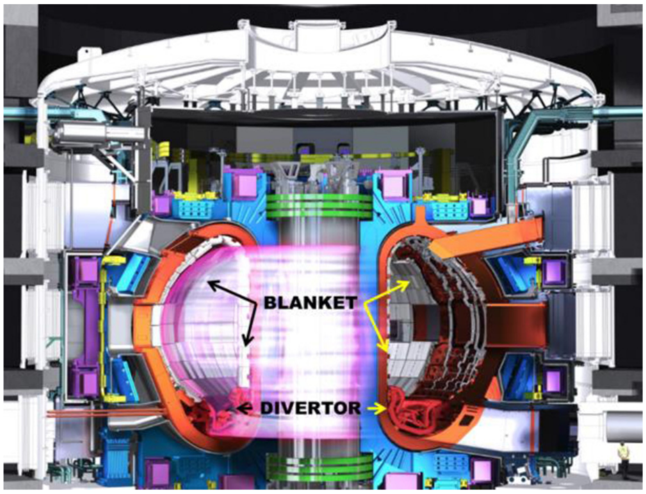

Fig. 1. Location of the Blanket System and Divertor in ITER.

from the blanket. The coolant is routed firstly through the FW, and then through the SB. The BMs are segmented into 18 poloidal locations: rows 1–6 are the inboard region, rows 7–10 are the upper region and rows 11–18 are the outboard region [4].

The inboard and upper modules (except BM10) are segmented toroidally into 18 equal modules, and the outboard modules (except BM14 and 15) are segmented into 36 modules. In the upper and equatorial port region (BM10, 14 and 15), the modules are located between ports and therefore segmented into 18 modules. In the Neutral Beam (NB) area, vessel sectors 2, 3 and 4 have a custom segmentation for BM 14 and 15. The NB region with the neutral beam injection ports creates a particular geometry challenge for the blanket design and special modules have been developed. The NB shine-through also needs to be accommodated by the impacted BMs.

The pre-pulse and pulse coolant parameters at the Blanket inlet (that is, at the chimney bulk head) are:

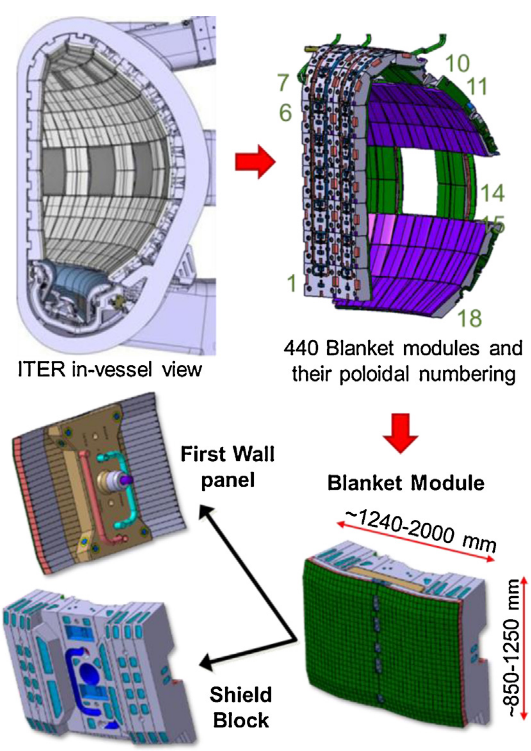

Fig. 2. The Blanket System.

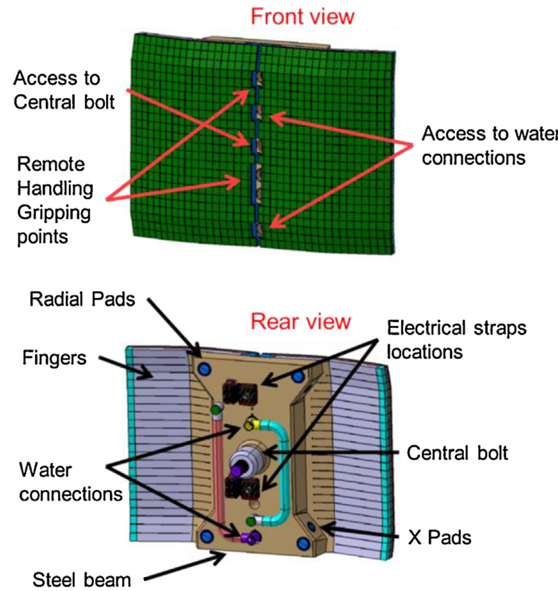

Fig. 3. First Wall panel of Blanket Module 1.

Inlet operating temperature: 70 °C (−5/+5 °C).

Inlet pressure: 4.0 MPa (−0.2/+0.6 MPa).

Mass flow rate for all 440 wall mounted Blanket Modules: a minimum of 3140 kg/s.

The requested measurement accuracy for the pressure, temperature and flow rate of the cooling water system is ±2% of the nominal value.

The Blanket shall be baked at 240 ± 10 °C by circulating hot water. During baking conditions, the nominal inlet pressure for the Blanket shall be 4.4 MPa at 10% of the total flow rate.

The design of the Blanket System was carried out by the Blanket Integrated Product Team, which include the ITER Organization and six Domestic Agencies (CN, EU, KO, JA, RF and US). This effort culminated in the Blanket Final Design Review held in April 2013 and formally closed in July 2013. The successful achievement of this milestone has allowed the Blanket System to progressively move towards the construction phase.

2.2. The First Wall panels

Following the ITER Design Review of 2007, the main functions of the Blanket System have been substantially expanded and it has now also to provide limiting surfaces that define the plasma boundary during startup and shutdown. This has led to a redefinition of the design heat fluxes and a shaping of the plasma facing surface to avoid the exposure of leading edges. The FW panels, which face the plasma, have become fully remote-handleable parts to ensure the possibility of replacing them in situ in case of damage [5].

The FW is shaped for minimizing the plasma heat load on edges caused by ports, diagnostic openings and assembly gaps between the FW panels. Similarly, shaping would help avoid leading edges in the case of radial misalignment of adjacent FW panels.

The design of the FW panel is structured on a strong backing steel beam, oriented in the poloidal direction (as shown in Fig. 3). The beam section is typically 350 mm (toroidal) × 150 mm (radial), over the entire poloidal length of the FW. Elongated plasma facing units, called "fingers" are attached to the beam in the toroidal direction with an overhang for full coverage of the SB. The fingers are constructed using a composition of three main materials, 316L(N)-IG austenitic steel for the structure, copper chromium zirconium (CuCrZr) alloy for the heat sink and beryllium as the plasma-facing material (or "armour").